NSP

Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Welch, Minnesota 55089

July 6, 1998

10 CFR Part 50
Section 50.73

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

LER 1-98-08:
Reactor Trip Initiated by a Negative Flux Upon Control Rod Insertion Following Failure of Control Rod Drive Cable

The Licensee Event Report for this occurrence is attached. In the report, we made no NRC commitments. This event was reported via the Emergency Notification System in accordance with 10 CFR Part 50, Section 50.72, on June 5, 1998. Please contact us if you require additional information related to this event.

Joel P. Sorensen

Joel P. Sorensen
Plant Manager
Prairie Island Nuclear Generating Plant

c: Regional Administrator - Region III, NRC
NRR Project Manager, NRC
Senior Resident Inspector, NRC
Kris Sanda, State of Minnesota

Attachment

1161

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Prairie Island Nuclear Generating Plant Unit 1	05000 282	1 OF 6

TITLE (4)

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	05	98	98	-- 08	--	07	06	98	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1

POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	√	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jeff Kivi	612-388-1121

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	AA	CON	W120	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	√ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On June 5th, 1998, Prairie Island Nuclear Generating Plant (PINGP) Unit 1 was at steady-state, 100% power with the Rod Control System in automatic. At approximately 7:00 PM, Unit One tripped from 100% power on a Negative Flux Rate Trip generated by Power Range Nuclear Instrumentation. Subsequent investigation revealed that Control Rod G7 dropped because of equipment failure, and caused the Negative Flux Rate Trip.

Efforts to establish the equipment failure mechanism that caused the trip are ongoing.

EVENT DESCRIPTION

On June 5th, 1998, Prairie Island Nuclear Generating Plant (PINGP) Unit 1 was at steady-state, 100% power with the Rod Control System[1] in automatic. At approximately 7:00 PM, Unit One tripped from 100% power on a Negative Flux Rate Trip generated by Power Range Nuclear Instrumentation. Subsequent investigation revealed that Control Rod G7 dropped because of equipment failure, and caused the Negative Flux Rate Trip.

Early in the reactor trip recovery procedure, the reactor coolant system[2] (RCS) was cooling down and the operators reduced auxiliary feedwater[3] (AF) flow in accordance with procedure. However, the RCS temperature continued to decrease. Approximately six minutes after the trip, the turbine building operator reported that there was steam in the turbine building. Based on this information and the fact that the RCS was still cooling down, the shift supervisor ordered both main steam isolation valves[4] (MSIVs) closed. This action was in accordance with guidance in the reactor trip recovery procedure. A short time later, the turbine building operator reported the source of steam to be a lifted relief valve[5] on the tube side of the 15A feedwater heater[6] (the high pressure feedwater heater). The operating main feedwater pump[7] was stopped to reduce the tube side pressure and the relief valve reseated.

With the MSIVs closed the RCS cooldown was stopped and RCS temperature started trending upward. Because the MSIVs were closed, decay heat removal was being accomplished by using AF pumps and steam generator[8] (SG) power operated relief valves[5] (PORVs) in auto with a setpoint to control steam generator pressure at 1050 psig. (RCS temp 551 degrees F).

Approximately two hours after the trip, steam generator levels were approaching the narrow range level upper operating limit of 38%. This narrow range operating level requirement was imposed in late 1997 as a result of a review of an analysis of containment pressure response to a main steam line break and is significantly lower than the previous level requirement. The operator was observing normal SG level 'swells' (a few percent) each time the PORVs opened in automatic. These swells added to the indicated level in the SGs and decreased the margin to the narrow range level upper operating limit. To eliminate this variation in SG level, the operator placed the SG PORV controllers[9] in manual to establish a constant steam demand. Other control room personnel were not aware of this transfer to manual

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[1] (EIIS System Identifier: AA)
[2] (EIIS System Identifier: AB)
[3] (EIIS System Identifier: BA)
[4] (EIIS Component Identifier: ISV)
[5] (EIIS Component Identifier: RV)
[6] (EIIS Component Identifier: HX)
[7] (EIIS Component Identifier: P)
[8] (EIIS Component Identifier: SG)
[9] (EIIS Component Identifier: PCO)

control. As 12 SG narrow range level approached its upper operating narrow range level limit, the operator became focused on attempts to stop the level increase and did not realize that the manually established steam load was not sufficient to prevent an increase in RCS temperature and SG pressure. About ten minutes after the PORVs were placed in manual control, SG pressure increased to the point that a SG safety valve[5] on 11 SG lifted. The opening of the safety valve caused SG pressure to drop and 11 SG narrow range level to swell. Actual level did not exceed the narrow range design basis limit of 40.9%. Plant response was as expected. The operator subsequently stabilized SG levels and returned both PORV controllers to automatic.

CAUSE OF THE EVENT

Initial troubleshooting by I&C following the trip revealed a blown fuse[10] indicator on the positive side (+) fuse FU26 of Rod G7 (Control Bank C Group 2) in Power Cabinet 2AC. No other abnormal indications were observed. Using a digital multi-meter (DMM), I&C confirmed that Fuse FU26 was open and there was continuity for the return side (-) Fuse FU30.

Further troubleshooting by Plant Electricians commenced on June 6, 1998. The field leads for rod G7 stationary gripper were meggered and found to have a low impedance path (short) to ground. Subsequent troubleshooting revealed the short to be under the reactor missile shield at the reactor head. It was determined that removal of the Reactor Missile Shield was necessary, and Operations commenced a cooldown of the RCS. While cooling down, megger and continuity readings were taken for all 29 control rods. Lift coils for two additional rods (F12 & K5) were found to have megger readings lower than expected. Results of the continuity check of all rods were normal. Upon reaching Cold Shutdown and after removal of the head, megger and continuity readings were repeated with the Reactor Coolant System cold. The rod G7 megger readings did not change, but all other megger readings remained high or improved. Megger values of the lift coils for rods F12 and K5 improved to the same high insulation resistance readings as the other rods. The cables[11] to all rods were then disconnected at the head, and wire-to-wire megger readings taken for wire pairs associated with each coil (stationary gripper, movable gripper, and lift coil). The rod G7 cable was the only cable to show any low readings: the stationary gripper coil wires read low resistance, the remaining pairs of wires read greater than 200M-Ohms.

A previous trip occurred on Unit One in June 1997, where a faulted cable caused a negative rate trip. An independent analysis was inconclusive as to the apparent cause for the connector failure in 1997. With the recently failed cable, Westinghouse, the cable vendor, was contacted to perform a root cause

[10] (EIIS Component Identifier: FU)
[11] (EIIS Component Identifier: CBL)

analysis. The three cables (faulted G7, F12 and K5 with low megger readings) were collected from the head area and sent to Westinghouse for initial failure analysis. Westinghouse confirmed the Prairie Island findings described above. Westinghouse reported two other utilities have found similar degradation of rod control cables, but no shorts or other electrical problems resulted.

Based on preliminary discussion of the above findings with Westinghouse, the head area cables and connectors[12] were inspected. The 1997 and 1998 trips each were due to a connector failure in one of the five center core rods (rods H8 and G7, respectively). The connectors of the inner core rods were visually inspected. There was no evidence of corrosion, arcing, burning, or other failure initiators identified.

All CRDM cables were inspected. The material condition of the outer Hypalon cable jackets vary with core location. The Hypalon cable jacket is a protective covering and is not required for electrical insulation. The center core location cables have a coating of dust on the exterior of the jackets that appears to be a byproduct of degradation of the cable jacket. Furthermore, jackets with the dust coating exhibit loss of flexibility and embrittlement commensurate with the concentration of dust. The outer core location cables have little or no such dust coating. Portions of the cable that run from the head outside the missile shield to the junction box on the reactor cavity wall appear unaffected. The Tefzel electrical insulating material on individual conductors from dissected cables was inspected for any degradation. In all cases the Tefzel insulation appeared shiny, pliant, and flexible. The short circuit does not appear to be a result of a failure of the Tefzel insulating material.

Three out of the five center core rod control cables were replaced (locations G7, F6 & H6). The cable and connector to rod G7 was replaced because it was faulted; rod control cables and connectors to core locations F6 and H6 were also replaced as a precautionary measure because heat may be a factor in the failure. The remaining two center core location rod control cables G8 and H8 were not replaced because they were replaced following the 1997 trip: both cables, with one year of service, were inspected and found to have retained their flexibility and pliability, showing none of the usual characteristic signs of degradation (i.e. dust formation , embrittlement, etc.). The cables to outer core rods F12 and K5 were replaced as a precautionary measure due to the low initial megger readings found during the original troubleshooting efforts. Furthermore, to ensure that no external water could infiltrate into the cable connector, self-sealing Silicon tape was applied to the cable jacket-connector interface. The five new cables were meggered wire-to-wire, wire-to-ground, and checked for continuity; the results were satisfactory. All twenty-nine (29) cables were then re-meggered and continuity measurements were re-taken. The cables tested satisfactorily, again.

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[12] (EIIS Component Identifier: CON)

In the Rod Control System Power Cabinets, Firing Control Circuits, coil voltages and Urgent Alarm States were tested by I&C to ensure proper operation of the rod control system. The Rod Control System Power Supply preventative maintenance procedure was also performed to ensure the cable failure and subsequent short circuit did not damage the power supplies. The refueling Control Rod Drive Mechanism Timing procedure was performed twice to verify proper operation of the rod control system; once prior to replacing the reactor vessel missile shield and once at Hot Shutdown. The results of all tests were satisfactory, with no abnormalities.

Laboratory attempts to identify the failure mechanism are not, at this time, conclusive. There is evidence of material decomposition of the Hypalon jacket by an undetermined mechanism. A preliminary failure report from Westinghouse indicates that most of the elements found are attributable to materials used in the construction of the cable and connector and suggested that water intrusion together with the dust particles may have led to the fault between stationary gripper coil leads and/or ground. Whereas interaction between the dust particles and liquids possibly could have initiated the short inside the connector, Westinghouse has provided no evidence supporting a theory of intrusion of external moisture into the connector (for example, past sealing o-ring, between conductors inside the cable or through the viton insert) nor could PINGP technical staff provide any physical evidence in support of that theory.

Operator actions taken after the trip to control SG level by placing PORV controllers in manual are also being evaluated. Preliminary analysis of this event by the PINGP Error Reduction Task Force (ERTF) has identified three apparent inappropriate actions:

1. SG levels were not adequately controlled to maintain them within the specified operating range.

2. Transfer of the PORV controllers from 'auto' to 'manual' was not adequately communicated to the other Unit 1 Control Room personnel.

3. The process parameter being controlled by the PORV controller in 'auto' was not adequately monitored after the controllers were transferred to 'manual.'

The causes of these actions are being evaluated.

ANALYSIS OF THE EVENT

A short circuit inside the cable connector of control rod G7 at the CRDM caused the CRDM supply side fuse to blow open and the gripper released rod G7, allowing it to drop into the core. The Power Range

Nuclear Instrumentation System detected the negative flux rate and appropriately initiated the reactor trip.

This event is reportable under 10CFR50.73(a)(2)(iv) as an unplanned actuation of the RPS. The health and safety of the public were unaffected since the plant systems responded as designed to the automatic trip.

CORRECTIVE ACTION

PINGP reactor trip recovery procedures have been improved via temporary memo. Additional corrective actions to be taken include:

1. Complete analysis of the failure mechanism of failed connector.

2. To improve reliability, U1 and U2 CRDM cables in the affected area will be replaced with higher temperature rated jackets and stainless steel connectors.

3. The PINGP Error Reduction Task Force is investigating the steam generator level transients during the post trip recovery. The results of this investigation will be evaluated to identify any necessary corrective actions.

FAILED COMPONENT IDENTIFICATION

CRDM cable assembly to Rod G7. The cable assembly was supplied by Westinghouse (Part No. 8249C08G03).

PREVIOUS SIMILAR EVENTS

A trip due to a faulted head area cable assembly has been reported previously. Refer to Unit 1 Licensee Event Report No. 1-97-8, Unit One Reactor Trip June 2, 1997 due to faulted head area cable assembly.

CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9807140064 DOC.DATE: 98/07/06 NOTARIZED: NO DOCKET #
FACIL:50-282 Prairie Island Nuclear Station, Unit 1, Northern Stat 05000282

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SUBJECT: LER 98-008-00:on 980605,unit 1 reactor trip was noted.Caused by equipment failure.PINGP reactor trip recovery procedures have been improved via temporary memo.W/980706 ltr.

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